10 CFR 50.54(f)

Progress Energy

Serial: RNP-RA/05-0088

SEP 01 2005

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, Maryland 20852

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

RESPONSE TO NRC GENERIC LETTER 2004-02, "POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS"

Ladies and Gentlemen:

On September 13, 2004, NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," was issued requesting that licensees provide the requested information within 90 days and additional information by September 1, 2005. Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), is hereby providing the information requested by September 1, 2005, for H. B. Robinson Steam Electric Plant, Unit No. 2, in Attachment II to this letter.

Attachment I provides an Affirmation in accordance with the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f).

If you have any questions concerning this matter, please contact Mr. C. T. Baucom at (843) 857-1253.

Sincerely,

Jan F. Lucas

Jan F. Lucas
Manager – Support Services – Nuclear

CTB/cac

A116

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

Attachments:

I. Affirmation

II. Response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors"

c: Dr. W. D. Travers, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspector

AFFIRMATION

The information contained in letter RNP-RA/05-0088 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 01 September 2005

J. W. Moyer
J. W. Moyer
Vice President, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RESPONSE TO NRC GENERIC LETTER 2004-02, "POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PRESSURIZED-WATER REACTORS"

On September 13, 2004, NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," was issued requesting that licensees provide the requested information by September 1, 2005. Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. (PEC), is hereby providing the information requested by September 1, 2005, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, as follows:

NRC Request for Information 2(a):

Confirmation that the ECCS [emergency core cooling system] and CSS [containment spray system] recirculation functions under debris loading conditions are or will be in compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. This submittal should address the configuration of the plant that will exist once all modifications required for regulatory compliance have been made and this licensing basis has been updated to reflect the results of the analysis described above.

Response to 2(a):

The General Design Criteria (GDC) in existence at the time HBRSEP, Unit No. 2, was licensed for operation (July 1970) were contained in the proposed Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants," published in the Federal Register on July 11, 1967. HBRSEP, Unit No. 2, conformance with the Proposed GDC is described within Updated Final Safety Analysis Report (UFSAR) Section 3.1, "Conformance with General Design Criteria."

Generic Letter (GL) 2004-02 lists the applicable GDCs as follows: GDC 35, Emergency Core Cooling, GDC 38, Containment Heat Removal System, and GDC 41, Containment Atmosphere Cleanup. The comparable GDCs for HBRSEP, Unit No. 2, include GDC 41, GDC 44, and GDC 52, which state:

> GDC 41, Engineered Safety Features Performance Capability: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

> GDC 44, Emergency Core Cooling System Capability: An Emergency Core Cooling System with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit

the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

GDC 52, Containment Heat Removal System: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component.

Additionally, the Title 10 Code of Federal Regulations (10 CFR) listed in GL 2004-02, which include 50.46, 50.67, and Part 100, are also applicable to HBRSEP, Unit No. 2, as described in the UFSAR.

The containment sump recirculation functions under debris loading conditions will be in compliance with the applicable regulatory requirements based on the improved analyses and completion of the proposed modifications for the containment sump.

HBRSEP, Unit No. 2, has one common ECCS sump with a flat surface screen area of 116 square feet. The design of this sump is considered susceptible to clogging with a thin bed of fiber and loss-of-coolant accident (LOCA) generated particulate debris. Design and programmatic changes will be utilized to resolve Generic Safety Issue (GSI)-191 and GL 2004-02 head loss issues.

The strategy for resolution of GSI-191 and GL 2004-02 head loss issues includes the following basic features:

- Ensuring sufficient water supply reaches the containment sump during long term recirculation. This design constraint is accomplished by ensuring credited flow paths to the sump remain clear and by utilizing the minimum credible water level at the initiation of recirculation for design of the maximum height of the new sump screens.

- Minimizing head loss due to debris accumulation at the sump screens and improving the available net positive suction head by increasing surface area utilizing complex strainer geometry, providing adequate debris mass capture (interstitial volume) without impacting effective strainer surface area, and revising the containment insulation program to ensure that insulation changes improve the material characteristics from a head loss perspective.

- Minimizing latent debris by maintaining containment close-out cleanliness, foreign material exclusion standards, and an effective coatings program.

NRC Request for Information 2(b):

A general description of and implementation schedule for all corrective actions, including any plant modifications, that you identified while responding to this generic letter. Efforts to implement the identified actions should be initiated no later than the first refueling outage starting after April 1, 2006. All actions should be completed by December 31, 2007. Provide justification for not implementing the identified actions during the first refueling outage starting after April 1, 2006. If all corrective actions will not be completed by December 31, 2007, describe how the regulatory requirements discussed in the Applicable Regulatory Requirements section will be met until the corrective actions are completed.

Response to 2(b):

Analyses of debris generation, debris transport, and head loss for the proposed sump screen design have been completed for HBRSEP, Unit No. 2, based on the methodology presented in Nuclear Energy Institute (NEI) guidance report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, and the associated report titled, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004." The analyses, which are discussed further in item 2(c), were utilized to generate a design that will accommodate the most limiting post-LOCA debris generation and accumulation condition, with margin available to address chemical effects phenomena currently under evaluation. The proposed design is considered preliminary. The design will be further validated and adjusted, as necessary, as it is finalized. The values provided below for head loss and available net positive suction head (NPSH) margin associated with the preliminary design are not expected to change substantially during the finalization of the design.

The existing containment sump (also called the ECCS sump), which has an approximate overall area of 116 square feet (sq ft), will be replaced with a new sump screen with a minimum surface area of 3000 sq ft. The new screen will have a complex geometry, which minimizes thin bed effects, with perforations of 0.125 inch. This perforation size was selected to preclude blockage of downstream piping components. The perforation size of the replacement screen is smaller than that of the existing screen. The limiting components with respect to downstream component blockage are the CSS and residual heat removal (RHR) pump mechanical seal coolers. The tubing in these coolers is 1/4 inch in diameter with a free flow hole of 0.160 inch. If the downstream wear evaluation currently in progress identifies components with unacceptable wear that can be mitigated by smaller screen openings, the final design will incorporate those results. The larger screen surface area is needed to accommodate the maximum postulated quantity of accident-generated and latent debris that could reach the sump screen. The larger screen ensures that adequate NPSH is maintained under debris-laden conditions.

The existing coarse mesh screens located at the base of the primary shield wall, which separate the reactor coolant loop compartments from the outer containment areas, will be removed. These coarse mesh screens, if left in place, could become blocked with LOCA-generated debris causing the hold-up of water in the loop compartments that is needed in the outer area for ECCS recirculation.

Floor drains in four locations are relied upon to drain containment areas during post-LOCA mitigation. Trash racks will be installed around these drains.

It is currently planned that the analyses (including evaluation of downstream effects as discussed in response to 2(d)(vi)) will be completed and the design changes will be approved by

November 7, 2006. The proposed modifications will be installed by May 30, 2007. Any additional corrective actions that may be required will be completed by December 31, 2007.

NRC Request for Information 2(c):

A description of the methodology that was used to perform the analysis of the susceptibility of the ECCS and CSS recirculation functions to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. The submittal may reference a guidance document (e.g., Regulatory Guide 1.82, Rev. 3, industry guidance) or other methodology previously submitted to the NRC. (The submittal may also reference the response to Item 1 of the Requested Information described above. The documents to be submitted or referenced should include the results of any supporting containment walkdown surveillance performed to identify potential debris sources and other pertinent containment characteristics.)

Response to 2(c):

As previously stated, analyses of debris generation, debris transport, and head loss for the proposed sump screen design have been completed for HBRSEP, Unit No. 2, based on the methodology presented in Nuclear Energy Institute (NEI) guidance report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, and the associated report titled, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Revision 0, December 6, 2004." The analysis to support the preliminary design of the ECCS sump modification for HBRSEP, Unit No. 2, was performed by an alliance of the Enercon, Alion, and Westinghouse Corporations.

The minimum post-LOCA containment steady-state water level was determined based on limiting break location and size, which was assumed to occur at the high point in the reactor coolant system (RCS). The calculation of containment water level includes modeling of fluid held-up or not available, as discussed in the response to item 2(d)(iv).

The result of the water level calculation shows that the most limiting case, which is the small break LOCA of the pressurizer spray line, results in a depth of at least 1.2 feet above the containment floor, and the applicable emergency operating procedures ensure that at least 1.5 feet is available at the start of containment sump recirculation. The ECCS sump design will be such that the water level will completely submerge the screens.

Containment walkdowns were conducted and documented based on the guidance provided in NEI 02-01, Revision 1, "Condition Assessment Guidelines: Debris Sources Inside PWR Containments, September 2002." The results of the containment walkdowns provided input to the recirculation flow path assessment and development of a source term for a debris generation calculation. The purpose of the recirculation flow path assessment was to identify physical and structural features that could affect the flow of debris and water from a potential break location to the sump.

Areas of concern identified by the walkdowns include:

- The coarse mesh screens located at the base of the primary shield wall could become blocked with LOCA-generated debris causing the hold-up of water needed for ECCS recirculation.

- Various floor drains, relied upon to drain containment areas during post-LOCA mitigation, could become clogged with debris.

Insulation quantities from the containment walkdowns and plant design information were used in the debris generation calculation. In this calculation, break locations were considered in accordance with Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," November 2003. These break locations were evaluated using a three-dimensional computer model of the containment with material-specific zone of influence (ZOI) in accordance with NEI 04-07 methodology and the NRC Safety Evaluation Report (SER). A destruction pressure corresponding to 2.4 psig or a ZOI radius of 28.6 pipe inside diameters was assumed for Asbestos, Unibestos, Unspecified Fiberglass, and Kaylo. Additionally, these materials are assumed destroyed as 100% fines. There is no available destruction test data on these materials at this time. The 2.4 psig destruction pressure is the lowest value identified in the Table 3-2 of the NRC SER for the NEI 04-07 report. The associated ZOI involves the entire pump bay; therefore, the use of these values is considered conservative.

Once mapped, the ZOI is used with the piping isometrics to identify the debris generation for each break location. The debris generation is then assigned size distributions and defined by material characteristics. A latent debris source term of 400 lbs (340 lbs particulate and 60 lbs fiber) was assumed. This assumption will be validated by December 1, 2005, as previously stated in the 90-day response to NRC Generic Letter 2004-02, by letter dated March 4, 2005.

The identified debris source term was used as an input to the debris transport calculation. The methodology utilized to determine the amount of debris transported is based on the methodologies described in NUREG/CR-6762, Volume 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," and Section 3.6.3.1 of the NRC SER associated with NEI 04-07. The debris transport analysis considers each type of debris and the size of debris. The debris size fractions for the destruction of specific types of debris, which are described in the NRC SER associated with NEI 04-07, include the erosion of large pieces to small fines.

The results from the debris transport calculation were used as an input to the strainer head loss calculation. This calculation shows that the debris-bed head loss across a 3,000 sq ft strainer with a total flow of 3,820 gpm at 212 degrees F (sump fluid temperature) in recirculation mode is 1.49 ft-water, and at 209 degrees F the head loss is 1.52 ft-water. This result is based upon the limiting large break LOCA-generated debris bed, as discussed previously.

An analysis was completed to verify that two-phase flow does not occur due to pressure drop across the sump screen. This analysis confirmed that the minimum pressure will remain above the saturation pressure for the sump fluid. This analysis included the additional subcooling margin gained by accounting for the partial pressure of air in the steam-air mixture in the containment atmosphere during a LOCA. The partial pressure of air was not used in the determination of NPSH. It was only used in the verification that the proposed sump screen design would not be subject to an unacceptable amount of two-phase flow.

Flat bed strainers utilized in testing demonstrate a thin bed effect, where the particulate causes a large head loss when combined with a relatively small accumulation of fiber (i.e., approximately 1/8 inch thick). The proposed strainer design is a complex geometry design, which is not expected to be subject to uniform debris accumulation. At 3,820 gpm and with a 3,000 sq ft strainer area, the approach velocity is approximately 0.003 feet per second (fps).

In order to estimate the potential thin bed effect, a calculation assuming a 3/8 inch thin bed thickness was conducted. The result showed that the head loss would be 1.43 ft-water. The thin bed thickness of 3/8 inch was chosen based upon engineering judgment that the complex strainer design will not exhibit any significant head loss until at least 3/8 inch in equivalent fiber by volume is present. The computer code utilized indicated that the solidity fraction limit was exceeded. Therefore, the calculated thin bed head loss will be validated through flow rate testing (applicable to HBRSEP, Unit No. 2) by November 7, 2006. If the thin bed flat screen head loss testing shows unacceptable results, then strainer-specific testing will be completed by November 7, 2006, to validate that the complex strainer design for HBRSEP, Unit No. 2, will not be subject to the thin bed effect with less than 3/8 inch equivalent volume of fiber.

The debris quantities present at the ECCS sump screen include large amounts of particulate. Testing at the Los Alamos National Laboratory (LANL) was conducted with the mass ratio of particulate-to-fibrous debris varied from 0.5 to 2.0. However, in the large break LOCA case for HBRSEP, Unit No. 2, this ratio is estimated to be about 10. Section 3.7.2.3 of the NRC SER associated with NEI 04-07 provides several caveats on the use of the NUREG/CR-6224 head loss correlation. Section 3.7.2.3.1.4 of the NRC SER states that the NUREG/CR-6224 head loss correlation can only be used as a scoping tool when calcium silicate is one of the debris bed constituents. Therefore, testing will be performed at flow rates and debris bed compositions that are representative for HBRSEP, Unit No. 2. This debris bed head loss testing will be completed by November 7, 2006.

The susceptibility of the ECCS and CSS recirculation downstream flow paths and components to LOCA-generated debris is being performed based on methods described in report WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," dated June 2005. This evaluation is currently in progress. The status of the downstream evaluations for blockage and wear are provided in items 2(d)(v) and 2(d)(vi) and were previously communicated by letter dated July 19, 2005, which was a response to the NRC request for additional information related to Generic Letter 2004-02. No corrective actions have been identified based on the evaluations completed to date. However, corrective actions may be identified during the course of completing this evaluation. The corrective actions will be addressed in approved design change documentation by November 7, 2006, with implementation by May 30, 2007.

NRC Request for Information 2(d)(i):

The minimum available NPSH margin for the ECCS and CSS pumps with an unblocked sump screen.

Response to 2(d)(i):

The safety injection (SI) and CSS pumps are aligned for containment sump recirculation cooling mode by use of the discharge of the RHR pumps. The most limiting NPSH condition for these pumps occurs during the injection phase of a LOCA when the suction of these pumps is aligned to the Refueling Water Storage Tank (RWST). The NPSH margin for the RHR pumps during containment sump recirculation, assuming an unblocked sump screen, is 3.3 ft-water.

NRC Request for Information 2(d)(ii):

The submerged area of the sump screen at this time and the percent of submergence of the sump screen (i.e., partial or full) at the time of the switchover to sump recirculation.

Response to 2(d)(ii):

The submerged screen area of the sump screen is based on a minimum of 3000 sq ft. The extent of submergence, at the time of switchover to sump recirculation, is 100%.

NRC Request for Information 2(d)(iii):

The maximum head loss postulated from debris accumulation on the submerged sump screen, and a description of the primary constituents of the debris bed that result in this head loss. In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) and CSS washdown should be considered in the analyses. Examples of this type of debris are disbonded coatings in the form of chips and particulates and chemical precipitants caused by chemical reactions in the pool.

Response to 2(d)(iii):

As discussed in the response to item 2(c), the debris inventory (result of design reviews and walkdowns) is utilized to determine a debris source term due to various LOCA break locations. The debris generation calculation has identified that a large break LOCA in the cross-over leg at the steam generator outlet within the loop 3 (also called "C" loop) reactor coolant pump compartment generates the maximum possible quantity of debris, while the same break in loop 2 ("B" loop) generates the largest particulate debris load (i.e., the largest particulate-to-fiber ratio). After applying the appropriate transport fractions to these limiting debris source terms, the head loss calculation determined that the limiting debris bed is caused by the larger particulate load, as compared to the larger volume of debris. The constituents of this limiting debris bed are summarized as follows:

- Latent Dirt, Dust, and Fiber
- Qualified Coatings
- Unqualified Coatings
- Cal-Sil/Asbestos
- Cal-Sil
- Kaylo

- Nukon®
- Temp Mat™
- Nukon® or Temp Mat™
- Temp Mat™ or Kaowool
- Unibestos
- Fiberglass
- Reflective Metal Insulation (RMI)

Pending quantification of the head loss increase due to additional industry chemical-effects testing, the proposed replacement screen size was increased to the extent practicable to maximize the margin available to accommodate potential chemical effects. This margin is equal to the clean-screen NPSH margin minus the debris bed head loss reported above (3.3 ft-water minus 1.52 ft-water, which equals 1.78 ft-water). This margin can accommodate up to a 117% increase in debris bed head loss due to chemical effects.

The NEI Sump Performance Task Force has generated a preliminary set of debris bed head loss adjustment factors based upon four Integrated Chemical Effects Tests (ICET). These adjustment factors were presented in the NEI Generic Letter 2004-02, September 1, 2005, response template. The four ICET cases were performed with either sodium hydroxide (NaOH) or tri-sodium phosphate (TSP). The two NaOH cases represent plants with either 100% fiberglass or 80% calcium silicate and 20% fiberglass.

HBRSEP, Unit No. 2, utilizes NaOH, injected via the containment spray system, to control sump pH in a range of 8.5 to 11, and has both fiberglass and calcium silicate debris. Therefore, HBRSEP, Unit No. 2, is judged to be best represented by ICET #4 (NaOH buffer and 80% cal-sil and 20% fiberglass insulation with a sump pH of 10). The maximum value of the proposed debris bed head loss adjustment factor for ICET #4 is 4%. Further examination of ICET results and follow-on head loss testing will be needed to provide the necessary confirmation of these factors; however, the available margin is expected to bound any increase in these factors.

As shown in the preceding list, HBRSEP, Unit No. 2, debris includes several insulating materials that are not represented by ICET. HBRSEP, Unit No. 2, will perform further evaluations and/or testing, as appropriate, to characterize the chemical effects associated with these insulating materials by November 7, 2006.

<u>NRC Request for Information 2(d)(iv):</u>

The basis for concluding that the water inventory required to ensure adequate ECCS or CSS recirculation would not be held up or diverted by debris blockage at choke-points in containment recirculation sump return flowpaths.

<u>Response to 2(d)(iv):</u>

As discussed in the response to item 2(c), the containment steady-state water level calculation was revised to address specific concerns identified by GSI-191 pertaining to water level. The revised calculation includes modeling of fluid held-up or not available, such as:

- Steam hold-up in the containment atmosphere
- Water volume required to fill the RHR and CSS piping that is empty prior to the LOCA
- Additional mass of water that must be added to the RCS due to the increase in the water density at the lower sump water temperatures (versus the RCS temperature prior to the LOCA)
- Condensation on surfaces
- Water volume required to fill the pressurizer steam space
- Water in transit from the CSS nozzles and the break to the containment sump
- Water hold-up in the refueling canal
- Water hold-up within the curbs in the reactor coolant pump platform
- Water lost through ECCS leakage from containment
- Miscellaneous hold-up volume including: small quantity of leakage into the Seal Table Room; small quantity of water that leaks past the water repellant mastic or metal jacketing covering piping and component insulation; small quantity of water that might hold-up in the containment building elevator; and, hold-up in the containment drainage piping

Further assurance that the assumed volume of water will reach the containment sump is based on the modifications to remove the coarse mesh screens at the base of the reactor coolant loop compartments and installation of protective trash racks around the floor drains, as discussed in the response to item 2(b).

The result of the calculation shows that the most limiting case, which is a small break LOCA of the pressurizer spray line, results in a depth of at least 1.2 feet above the containment floor. An additional conservatism of this calculation is that injection from the RWST continues beyond this time providing an additional 60,000 gallons of water to the containment building before injection is terminated. The applicable emergency operating procedures ensure a minimum water level of 1.5 ft above the floor at the start of containment sump recirculation. Therefore, there is sufficient water level in the containment building at the initiation of switchover to completely cover the proposed ECCS sump screen.

NRC Request for Information 2(d)(v):

The basis for concluding that inadequate core or containment cooling would not result due to debris blockage at flow restrictions in the ECCS and CSS flowpaths downstream of the sump screen (e.g., a HPSI [high pressure safety injection] throttle valve, pump bearings and seals, fuel assembly inlet debris screen, or containment spray nozzles). The discussion should consider the adequacy of the sump screen's mesh spacing and state the basis for concluding that adverse gaps or breaches are not present on the screen surface.

Response to 2(d)(v):

The downstream effects evaluations are in progress and have identified components that are potentially susceptible to blockage. The identified components are the SI pump internal flow passages and the reactor internals/fuel.

The schedule for completing the downstream effects evaluation and addressing any identified susceptibilities by December 31, 2007, is provided as follows:

- The downstream effects evaluations will be completed in accordance with the methodology of WCAP-16406 by December 29, 2005, with the exception of reactor internals/fuel and SI pump internal flow passages.
- The downstream effects evaluations, performed in accordance with WCAP-16406-P, will be completed for the reactor internals/fuel and SI pump internal flow passages by July 13, 2006.
- Any modifications identified during the downstream effects evaluation will be finalized by November 7, 2006.
- Implementation of modifications identified during the downstream effects evaluation will be completed by May 30, 2007.

NRC Request for Information 2(d)(vi):

Verification that close-tolerance subcomponents in pumps, valves and other ECCS and CSS components are not susceptible to plugging or excessive wear due to extended post-accident operation with debris-laden fluids.

Response to 2(d)(vi):

As previously stated, the downstream effects evaluations are in progress. The components potentially susceptible to wear have been identified. The components identified that are potentially susceptible to wear include:

Pump internals:
SI pump wear rings and impellers
RHR pump wear rings and impellers
CSS pump wear rings and impellers

Pump seals:
SI pump seals
RHR pump seals
CSS pump seals

Evaluations completed thus far do not indicate a concern with excessive wear of the system piping, orifices, heat exchanger tubes, spray nozzles, or valves. Evaluations of pump clearances indicate the clearances will exceed the manufacturers' replacement specifications. However, flow performance is not expected to be adversely affected.

The schedule for completing the downstream effects evaluation and addressing any identified susceptibilities by December 31, 2007, is consistent with the schedule provided in the response to item 2(d)(v).

NRC Request for Information 2(d)(vii):

Verification that the strength of the trash racks is adequate to protect the debris screens from missiles and other large debris. The submittal should also provide verification that the trash racks and sump screens are capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under predicted flow conditions.

Response to 2(d)(vii):

The final design will ensure that trash racks are capable of protecting the associated debris screens from missiles and other large debris, and that the trash racks and sump screens are capable of withstanding the applicable design basis loads.

NRC Request for Information 2(d)(viii):

If an active approach (e.g., backflushing, powered screens) is selected in lieu of or in addition to a passive approach to mitigate the effects of the debris blockage, describe the approach and associated analyses.

Response to 2(d)(viii):

Active sump screen features are not being utilized for the planned modifications at HBRSEP, Unit No. 2.

NRC Request for Information 2(e):

A general description of and planned schedule for any changes to the plant licensing bases resulting from any analysis or plant modifications made to ensure compliance with the regulatory requirements listed in the Applicable Regulatory Requirements section of this generic letter. Any licensing actions or exemption requests needed to support changes to the plant licensing basis should be included.

Response to 2(e):

The licensing bases changes include the new analytical bases for the containment ECCS recirculation flow path and the associated plant modifications. The plant modifications (as described in the response to item 2(b)) do not trigger the 10 CFR 50.59 screening criteria, which are stated as follows:

- Does the proposed activity involve a change to a system, structure, or component (SSC) that adversely affects any FSAR-described design function?

- Does the proposed activity involve a change to a procedure that adversely affects how any FSAR-described SSC design function is performed or controlled?

- Does the proposed activity involve revising or replacing any FSAR-described evaluation methodology that is used in establishing the design bases or used in the safety analyses?

- Does the proposed activity involve a test or experiment not described in the FSAR, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the FSAR?

Therefore, the proposed modifications will not require further evaluation against the criteria described in 10 CFR 50.59(c)(2), and NRC approval is not required.

As stated in the response to item 2(c), the analytical methodology is based on NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," which was reviewed by the NRC and approved via Safety Evaluation Report dated December 6, 2004. Based on the analyses that have been completed for HBRSEP, Unit No. 2, it is judged that NRC review in accordance with 10 CFR 50.59(c)(2) will not be required. Further, the methodology being used does not alter the performance (i.e., flow rate and temperature) of the ECCS as currently analyzed in the UFSAR for accident mitigation. The analyses conducted in response to GL 2004-02 provide additional assurance that the ECCS will perform as required for accident mitigation when containment sump recirculation is needed.

The current analysis of the ECCS and the existing containment sump screen performance is based on the assumption of 80% blockage of the containment sump screen wetted surface area. This amount of blockage was determined to be insignificant in the determination of RHR pump NPSH when aligned for containment sump recirculation. The proposed calculation basis methodology for RHR pump NPSH is described in the response to item 2(c). The results provided in the response to item 2(d)(i) show that adequate NPSH margin is maintained.

As previously described in the response to item 2(c), an analysis to verify that two-phase flow does not occur due to pressure drop across the sump screen was completed. This analysis confirmed that the minimum pressure will remain above the saturation pressure for the sump fluid. This analysis included the additional subcooling margin gained by accounting for the partial pressure of air in the steam-air mixture in the containment atmosphere during a LOCA. The partial pressure of air was not used in the determination of NPSH. It was only used in the verification that the proposed sump screen design would not be subject to an unacceptable amount of two-phase flow. This analysis is considered part of the evaluation of containment sump performance and it is expected that this will be incorporated, as appropriate, into the design and licensing basis for HBRSEP, Unit No. 2.

A review of the existing requirements for HBRSEP, Unit No. 2, resulted in no identified exemptions or Operating License changes being required. Therefore, it is currently expected that no licensing actions will be required for the implementation of the changes associated with resolution of GSI-191.

<u>NRC Request for Information 2(f):</u>

A description of the existing or planned programmatic controls that will ensure that potential sources of debris introduced into containment (e.g., insulations, signs, coatings, and foreign materials) will be assessed for potential adverse effects on the ECCS and CSS recirculation functions. Addressees may reference their responses to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," to the extent that their responses address these specific foreign material control issues.

Response to 2(f):

Specification L2-M-039, Revision 2, "Piping and Equipment Thermal Insulation," contains the requirements for insulation in the plant and also specifically addresses the containment insulation requirements. Specification L2-M-039 describes the insulating materials currently available and acceptable. This specification includes a section specifically for insulation inside containment, which considered the regulatory positions described in Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling following a Loss-of Coolant Accident." The thermal insulation for replacement inside containment is limited to metal reflective insulation and fiberglass blankets. Changes to insulation inside containment are required to be recorded and submitted to HBRSEP, Unit No. 2, Engineering.

Walkdowns of the HBRSEP, Unit No. 2, containment were performed and an insulation inventory was generated. This inventory is considered the baseline and was used in the debris generation analysis. The requirements for containment insulation will maintain the insulation design basis used in the debris generation analysis. Insulation changes are recorded, a copy attached to the applicable plant maintenance records, and a copy sent to Engineering in accordance with the specification. Deviations from the specification require review and evaluation by Engineering using the debris generation analysis inventory as the design basis.

The maintenance work instructions include the requirement that insulation is installed in accordance with Specification L2-M-039. This requirement is included in procedure MMM-003, Revision 73, "Maintenance Planning." Existing preventive maintenance work instructions are also being revised to include these instructions. Insulation installed in the plant will be maintained in accordance with the debris generation analysis through the use of these procedural and specification controls.

Programmatic control of containment general cleanliness and the condition of non-insulating materials (e.g., coatings) has been previously described in response to NRC Bulletin 2003-01, under the discussion of compensatory measures 4 and 5. In general, the control of containment cleanliness is based on the use of program procedure PLP-006, "Containment Vessel Inspection/ Closeout," which provides instructions for complete and consistent closeout inspection of the containment, including removal of signs and foreign material, and restrictions on coatings. PLP-006 is normally completed at the last containment entry prior to RCS heatup after a major outage (e.g., refueling outage), or for any entry into the containment while RCS temperature is above 200 degrees F.

Additionally, HBRSEP, Unit No. 2, employs a coatings program as previously described in response to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and

Protective Coatings Deficiencies and Foreign Material in Containment." This program requires condition assessments to be conducted each refueling outage and unqualified/degraded coatings are replaced, removed, or assessed for potential impact on sump performance. The debris generation calculation utilized in the GL 2004-02 evaluations is based upon the current condition assessment of containment coatings, conducted in accordance with the coatings program procedures. This program will be updated to include the requirements associated with evaluations for GL 2004-02, after completion of the proposed modifications.